Joseph A. Widay
Plant Manager

1503 Lake Road
Ontario, New York 14519-9364
585.771.3000

Constellation Energy
R.E. Ginna Nuclear Power Plant, LLC

October 27, 2004

Mr. Robert L. Clark
Office of Nuclear Regulatory Regulation
U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Subject: Sixty (60) Day Response to Generic Letter (GL) 2004-01
Requirements for Steam Generator Tube Inspections
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Clark:

On August 30, 2004 the NRC issued GL 2004-01, Requirements for Steam Generator Tube Inspections. The purpose of the letter was to:

(1) *advise addressees that the NRC's interpretation of the technical specification (TS) requirements in conjunction with 10 CFR Part 50, Appendix B, raises questions as to whether certain licensee steam generator (SG) tube inspection practices ensure compliance with these requirements,*

(2) *request that addressees submit a description of the tube inspections performed at their plants, including an assessment of whether these inspections ensure compliance with the TS requirements in conjunction with 10 CFR Part 50, Appendix B,*

(3) *request that addressees who conclude they are not in compliance with the SG tube inspection requirements contained in their TS in conjunction with 10 CFR Part 50, Appendix B, propose plans for coming into compliance with these requirements, and*

(4) *request addressees to submit a tube structural and leakage integrity safety assessment that addresses any differences between their practices and the NRC's position regarding the requirements of the TS in conjunction with 10 CFR Part 50, Appendix B. A safety assessment should be submitted for all areas of the tube required to be inspected by the TS where flaws have the potential to exist and inspection techniques capable of detecting these flaws are not being used. This assessment should include an evaluation of whether the inspection practices rely on an acceptance standard different from the TS acceptance standards and whether the technical basis for these inspection practices constitutes a*

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change to the "method of evaluation" (as defined in 10 CFR 50.59) for establishing the structural and leakage integrity of the tube-to-tubesheet joint.

Pursuant to 10 CFR 50.54(f), addressees are required to submit a written response to the generic letter. The attachment to this letter provides the R.E. Ginna Nuclear Power Plant response.

If you should have any questions regarding this submittal, please contact Mr. Thomas Harding, 585-771-3384.

Very truly yours,

Joseph A. Widay

Joseph A. Widay

STATE OF NEW YORK :
: TO WIT:
COUNTY OF WAYNE :

I, Joseph A. Widay, being duly sworn, state that I am Vice President – R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC), and that I am duly authorized to execute and file this response on behalf of Ginna LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Ginna LLC employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Joseph A. Widay

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Monroe, this 27 day of October, 2004.

WITNESS my Hand and Notarial Seal:

Sharon L Miller
Notary Public

SHARON L. MILLER
Notary Public, State of New York
Registration No. 01MI6017755
Monroe County
Commission Expires December 21, 20 06

My Commission Expires:

12-21-06
Date

xc: Mr. Robert Clark (Mail Stop O-8-C2)
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

U.S. NRC Ginna Senior Resident Inspector

James M. Petro Jr., Esquire
Counsel
Constellation Energy
750 East Pratt Street, 5th Floor
Baltimore, MD 21202

Requested Information 1

Addressees should provide a description of the SG tube inspections performed at their plant during the last inspection. In addition, if they are not using SG tube inspection methods whose capabilities are consistent with the NRC's position, addressees should provide an assessment of how the tube inspections performed at their plant meet the inspection requirements of the TS in conjunction with Criteria IX and XI of 10 CFR Part 50, Appendix B, and corrective action taken in accordance with Appendix B, Criterion XVI. This assessment should also address whether the tube inspection practices are capable of detecting flaws of any type that may potentially be present along the length of the tube required to be inspected and that may exceed the applicable tube repair criteria.

Response:

Steam Generator tube inspections performed at the R.E Ginna Nuclear Power Plant (Ginna NPP) are consistent with the NRC's position regarding tube inspections.

Background

Ginna NPP is a two loop PWR and contains Babcock & Wilcox replacement steam generators (RSGs). Each steam generator contains 4765 thermally treated Inconel-690 U-tubes that have an outer diameter of 0.750-inches with a wall thickness of 0.043-inches. Secondary side support structures include eight 410 stainless steel lattice grids supporting the tube straight lengths, and up to twelve 410 stainless steel fan bars and collector bars supporting the U-bends depending on the tube radius. The tubing within the 25.25-inch tubesheet is hydraulically expanded throughout the full thickness of the tubesheet. The lower row U-bend region, Row 1 through Row 18 received additional thermal stress relief following the tube bending process. The Row 1 & Row 2 U-bends are crossover designed to maximize the bend radius and to minimize ovality. Ginna NPP operates on an approximate 18-month fuel cycle.

The Ginna NPP RSGs were installed in May 1996 and had operated for 5.0 effective full power years (EFPY) at the time of their last inspection in March of 2002. The 2002 outage was the fourth refueling outage since steam generator replacement.

Ginna Previous Inspection Information

The most recent Ginna NPP RSG tube inspection was performed in the March of 2002 refueling outage. This 2002 outage inspection scope was governed by: Ginna NPP Improved Technical Specification (ITS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program"; the Electric Power Research Institute (EPRI) Pressurized Water Reactor (PWR) SG Examination Guidelines Revision 5; ASME Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components,"1995 Edition, 1996 Addenda; (Steam Generator Management Program Activities); and the results of the Ginna NPP specific degradation assessment. The inspection techniques and equipment were capable of reliably detecting the specific degradation mechanisms applicable to the Ginna NPP RSGs. The inspection techniques, essential variables, and equipment were qualified to Appendix H, "Performance Demonstration for Eddy Current Examinations," of the EPRI PWR SG Examination Guidelines. These techniques were also site validated to Ginna NPP specific conditions.

The Ginna NPP March 2002 refueling outage RSG eddy current inspection scope included:

2002 "A" Steam Generator Scope

- 50% full-length bobbin coil inspection
- 20% +3"/-2" plus point / pancake coil MRPC inspection of top of tubesheet hot leg expansion transition
- 20% plus point Row 1 & Row 2 U-bend examination
- 100% full-length bobbin coil inspection of the periphery tubes with the possibility of having tube to tube proximity design tolerance deviation
- 100% plus point probe inspection of the bobbin coil confirmed proximity tubes (13 tubes).
- 20% plus-point probe inspection of hot leg dents and dings > 5.0 volts
- 20% plus point probe inspection of hot leg manufacturing buff marks > 5.0 volts
- *Plus-point probe inspection of all bobbin coil non-quantifiable indications (i.e., "I-codes")*
- X probe sample of indications to improve knowledge base (167 tubes)

2002 "B" Steam Generator Scope

- 50% full-length bobbin coil inspection
- 20% +3"/-2" plus point / pancake coil MRPC inspection of top of tubesheet hot leg expansion transition
- 20% plus point Row 1 & Row 2 U-bend examination
- 100% full-length bobbin coil inspection of the periphery tubes with the possibility of having tube to tube proximity design tolerance deviation
- 100% plus point probe inspection of the bobbin coil confirmed proximity tubes (12 tubes).
- 20% plus-point probe inspection of hot leg dents and dings > 5.0 volts
- 20% plus point probe inspection of hot leg manufacturing buff marks > 5.0 volts
- Plus-point probe inspection of all bobbin coil non-quantifiable indications (i.e., "I-codes")

Potential Damage Mechanisms

Prior to the steam generator inspections described above, Ginna NPP specific degradation assessments were developed. In addition to the ITS inspection requirements, the degradation assessments evaluated the EPRI PWR SG Examination Guidelines in effect at the time of the inspection and available industry data for steam generators of similar design to determine which potential damage mechanisms may exist in the steam generators. Potential damage mechanisms are defined as damage mechanisms that have occurred at Ginna NPP or have the potential to occur in the Ginna NPP steam generator tubing based on industry experience with similar designed SGs and SG tube material. Once the potential damage mechanisms were identified, qualified inspection techniques were used to inspect for the damage mechanisms in the respective areas.

The Ginna NPP degradation assessment performed prior to the 2002 refueling outage, identified the following as potential degradation mechanisms:

- Fan bar wear
- Lattice grid wear
- Foreign object wear
- Tube to tube contact wear
- Volumetric / sludge lance jet impingement

Ginna NPP has not detected any degradation to date, and has not plugged any tubes in the replacement steam generators during operation.

In addition to the potential degradation mechanisms, Ginna NPP has inspected additional areas of the tubing where the bobbin coil is not qualified, necessitating the use of specialty probes. Even though Ginna NPP operating experience and engineering analysis have assessed these areas as being extremely unlikely to have degradation, Ginna NPP has performed the exams to provide additional assurance, while improving the knowledge base. Specialty probes, such as the plus point, pancake, and X-probes are used. The specialty probes are used to inspect the 1) top of tube sheet secondary face expansion transitions, 2) low row u-bends, and 3) dings and dents. All other areas along the tube length can either be effectively inspected with the bobbin coil, or do not have a potential for degradation based on site specific and industry inspection results.

Prior to use, the inspection techniques were industry qualified and site validated, or for instances where site specific damage mechanism signals were not available, the techniques were validated and shown to be equivalent to the EPRI industry qualified techniques, in accordance with the requirements of the EPRI PWR SG Guidelines Appendix H. Use of these inspection techniques provided assurance that potential flaws that may have been present were identified and assessed against the applicable repair criteria.

The eddy current nondestructive testing examinations were performed by personnel qualified to the American Society of Mechanical Engineers (ASME) Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components,"1995 Edition, 1996 Addenda which invokes the use of CP-189, "Standard for Qualification and Certification of Nondestructive Testing Personnel," 1991 Edition and to the requirements of EPRI PWR SG Examination Guidelines Appendix G, "Qualification of Nondestructive Examination Personnel for the Analysis of Nondestructive Examination Data," (current revision). The nondestructive examination procedures and equipment used to perform the eddy current inspections met the requirements of the ASME Code Sections XI and V, "Nondestructive Examination," 1995 Edition, 1996 Addenda, as well as the requirements of the EPRI PWR SG Examination Guidelines (current revision). Ginna NPP procedures were in-place to verify and ensure that all personnel, equipment and inspection processes were qualified to appropriate requirements and that the examination results were reviewed and documented to assure that the test requirements were satisfied.

These measures ensure that the requirements of Ginna NPP ITS and 10 CFR Part 50, Appendix B Criteria IX, "Control of Special Processes," XI, "Test Control," and XVI, "Corrective Action," have been satisfied.

Requested Information 2

If addressees conclude that full compliance with the TS in conjunction with Criteria IX, XI and XVI of 10 CFR Part 50, Appendix B, requires corrective actions, they should discuss their proposed corrective actions (e.g., changing inspection practices consistent with the NRC's position or submitting a TS amendment request with the associated safety basis for limiting the inspections) to achieve full compliance. If addressees choose to change their TS, the staff has included in the Attachment suggested changes to the TS definitions for a tube inspection and for plugging limits to show what may be acceptable to the staff in cases where the tubes are expanded for the full depth of the tubesheet and where the extent of the inspection in the tubesheet region is limited.

Response:

Steam Generator tube inspections performed at Ginna NPP are consistent with the NRC's position in generic letter 2004-01 in regard to Technical Specifications in conjunction with Criteria IX, XI, and XVI of 10 CFR Part 50, Appendix B. Therefore, this item is not applicable and no corrective actions are required.

Requested Information 3

For plants where SG tube inspections have not been or are not being performed consistent with the NRC's position on the requirements in the TS in conjunction with Criteria IX, XI, and XVI of 10 CFR Part 50, Appendix B, the licensee should submit a safety assessment (i.e., a justification for continued operation based on maintaining tube structural and leakage integrity) that addresses any differences between the licensee's inspection practices and those called for by the NRC's position. Safety assessments should be submitted for all areas of the tube required to be inspected by the TS, where flaws have the potential to exist and inspection techniques capable of detecting these flaws are not being used, and should include the basis for not employing such inspection techniques. The assessment should include an evaluation of (1) whether the inspection practices rely on an acceptance standard (e.g., cracks located at least a minimum distance of x below the top of the tube sheet, even if these cracks cause complete severance of the tube) which is different from the TS acceptance standards (i.e., the tube plugging limits or repair criteria), and (2) whether the safety assessment constitutes a change to the "method of evaluation" (as defined in 10 CFR 50.59) for establishing the structural and leakage integrity of the joint. If the safety assessment constitutes a change to the method of evaluation under 10 CFR 50.59, the licensee should determine whether a license amendment is necessary pursuant to that regulation.

Response:

Steam Generator tube inspections performed at Ginna NPP are consistent with the NRC's position in generic letter 2004-01 in regard to Technical Specifications in conjunction with Criteria IX, XI, and XVI of 10 CFR Part 50, Appendix B. Therefore, this item is not applicable and no safety assessment is required.